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Abstract

SOME ENGINEERING STUDIES CONDUCTED IN INTRODUCING BRITISH TYPE REACTOR FOR THE TOKAI ATOMIC POWER STATION

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In connection with the Tokai Atomic Power Station, The Japan Atomic Power Company and Japan Atomic Energy Research Institute have jointly carried out extensive experimental studies on the engineering and safety aspects of the fuel elements and graphite.

Oxidation of punctured model fuel elements and graphite blocks in atmospheres of either carbon dioxide or air was investigated at various temperatures. Experiments were also undertaken to study the correlation of fission gas release with the extent of carbon dioxide oxidation of irradiated uranium. It was found that the amount of fission gas released during oxidation is roughly proportional to the extent of oxidation. The total amount of fission gas released throughout the oxidation period was, at 500°C, 11% of the initial fission gas content, at 600°C, 25% and at 700°C, 45%. The diffusion coefficients of Xe in the uranium oxides produced by the oxidation were measured as a supplementary experiment. In addition, the effect of carbon dioxide decomposition on the oxidation rate of uranium was also studied.

High temperature creep test were made on uranium in an attempt to study the collapse phenomenon of the fuel element. The secondary

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creep rates of various uranium were measured with a vacuum creep tester under constant loads of 2-3.5 kg/mm^2 at temperatures of 425-575°C. The plots of the creep rates as a function of the logarithm of the loads and of the receprocal of the temperature were found to be linear, and addition of iron or aluminum caused an increase in the creep resistance.

To study the neutron irradiation effect on the mechanical properties of Magnox AL 80, the dependence of the tensile strength on grain size and on deformation rate were investigated exprimentally and the result agreed with Petch's equation.

Since the fuel concerned is a hollow cylinder, collapse of the fuel might occur under reactor condition. To solve this problem, a reduced model of the fuel element was subjected to carbon dioxide pressures of 30-50 kg/cm^2 at temperatures of 450-550°C and thermally cycled between limits of $\pm$20°C. The results showed that the change in diameter were within experimental error and 0.05 mm/month reduction under the loads of less than 30 kg/cm^2 and of 40 kg/cm^2, respectively.

Heat flow characteristics of the fuel element were studied by means of a testing loop operating at 10 kg/cm^2 and at 800°C. A Magnox clad uranium fuel was internally heated in the loop and the heat conduction characteristics and hot spot coefficient were investigated by measuring the temperature distribution of the surface. The extent of deformation and the behavior during pressure reduction were also observed.

Mechanical strength testings were performed on the graphite block by whole piece and the test pieces cut out of the block. Various useful informations were obtained.